SUMMARY OF THE UNIVERSITY OF MISSOURI RESEARCH REACTOR HEU TO LEU CONVERSION FEASIBILITY STUDY

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ABSTRACT

The University of Missouri Research Reactor (MURR), the highest-powered University-owned research reactor in the U.S., operates at 10 MW$_{th}$ and is one of five U.S. high performance research and test reactors that are actively collaborating with the U.S. Department of Energy to find a suitable LEU fuel replacement for the currently required HEU fuel. A conversion feasibility study of using U-10Mo monolithic LEU fuel has been completed by MURR and ANL. We have concluded that the proposed LEU fuel assembly design in conjunction with an increased power level of 12 MW$_{th}$ will: maintain safety margins during operation; allow operating cycle lengths to be maintained for efficient and effective use of the facility; and preserve an acceptable level and spectrum of key neutron fluxes to meet the scientific mission of the facility.

Broad MCNP scoping studies in 2007 led to the selection of a proposed LEU fuel assembly design based on the following calculated parameters: power peaking factors, excess reactivity, and the fast and thermal fluxes available to the experimental facilities. Since then, detailed models have been developed to simulate the complex MURR fuel cycle for both HEU and LEU. REBUS-DIF3D has been used to perform depletion calculations. Resulting fuel compositions for limiting cores were analyzed with MCNP to determine three dimensional flux and power distributions, including the radial and axial impacts of critical rod positions and the azimuthal peaking effects in the under-moderated core. PLTEMP was applied to determine margin to flow instability for key HEU and LEU cores.

1.   INTRODUCTION

This paper summarizes the analyses performed to study the feasibility of converting the University of Missouri Research Reactor (MURR) current highly-enriched uranium (HEU) fuel to low-enriched uranium (LEU) fuel. Because of its compact core design (33 liters), which requires a much higher loading density of ^{235}U, MURR could not perform its mission with any previously qualified LEU fuel products. Indeed, in 1986, a BOLD VENTURE 3-D model, benchmarked against the only MURR destructively analyzed fuel element, was used to demonstrate that a silicide LEU core loaded to a density of 7.2 gU/cm^3, and with no fission product inventory, would result in a k-effective of less than 1.0 [1]. However, in 2006 with the prospect of the Global Threat Reduction Initiative (GTRI) fuel development program validating the performance of monolithic U-Mo foil fuels, MURR started actively collaborating with the GTRI-Conversion Program, and four other U.S. high-performance research and test reactors that use HEU fuel, to find a suitable LEU fuel replacement.

2.   GENERAL DESCRIPTION OF FACILITY, REACTOR & FUEL

The MURR is a multi-disciplinary research and education facility providing a broad range of analytical and irradiation services to the research community and the commercial sector [see www.murr.missouri.edu].   The MURR has six types of experimental facilities designed to support these services and research programs: the Center Test Hole (Flux Trap); the Pneumatic Tube System; the Graphite Reflector Region; the Bulk Pool Area; the (six) Beamports; and the Thermal Column. The first four (4) experimental facilities provide areas for the placement of sample holders or carriers in different regions of the reactor core assembly for the purposes of material irradiation. Some of the material irradiation services include transmutation doping of silicon, isotope production for the development of radiopharmaceuticals and other life-science research, and neutron activation analysis. The six beamports channel neutron radiation from the reactor core to experimental equipment which is used primarily to determine the structure of solids and liquids through neutron scattering and to perform Boron Neutron Capture Therapy (BNCT) experiments.

2.1   Basic Reactor Description

The MURR is a pressurized, reflected (beryllium and graphite), heterogeneous, open pool-type reactor, which is light-water moderated and cooled. The reactor is designed and licensed to operate at a maximum thermal power level of 10 MW with forced cooling, or up to 50 kW in the natural convection mode.

The reactor core assembly is located eccentrically within a cylindrically-shaped, aluminum-lined pool, approximately 10 feet (3.0 m) in diameter and 30 feet (9.1 m) deep. The reactor core consists of four major regions: central test hole (flux trap), fuel, control blade, and reflector. A two-dimensional view of the reactor core assembly is shown in Figure 1. The fuel region has
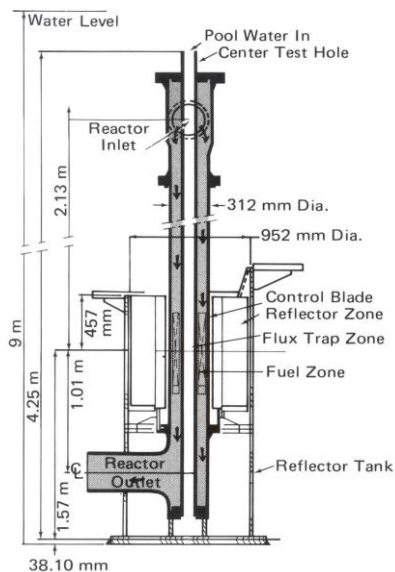

Figure 1
Reactor Core Assembly

a fixed geometry consisting of eight (8) fuel elements having identical physical dimensions placed vertically around an annulus between two cylindrical aluminum reactor pressure vessels. Each fuel assembly is comprised of 24 circumferential plates. The HEU plates contain uranium enriched to approximately 93% in the isotope ^{235}U as the fuel material. The control blade region is an annular gap between the outer pressure vessel and the inner reflector annulus, so that no penetration of the pressure vessels is required. Five (5) control blades operate vertically within this gap: four (4) Boral and one (1) stainless steel. The blades control the reactor reactivity by varying neutron reflection. The reflector region consists of two concentric right circular annuli surrounding the control blade region. The inner reflector annulus is a 2.71 inch (6.9 cm) thick solid sleeve of beryllium metal. The outer reflector annulus consists of vertical elements of graphite canned in aluminum, having a total thickness of 8.89 inches (22.6 cm).

2.2   Current Fuel Design and Operating Cycle

In 1971, MURR was converted from using the original uranium-aluminum alloy fuel to a uranium-aluminide dispersion UAl_x fuel material with a maximum loading of 775 grams of ^{235}U per element. The UAl_x dispersion fuel system was developed at the Idaho National Engineering Laboratory (INEL) for the high flux, high power Advanced Test Reactor (ATR) and subsequently used at the Materials Test Reactor (MTR) and Engineering Test Reactor (ETR) prior to its use at MURR [2, 3]. A drawing of the MURR fuel element is shown in Figure 2. Additional fuel element specifications can be found in Table 1.
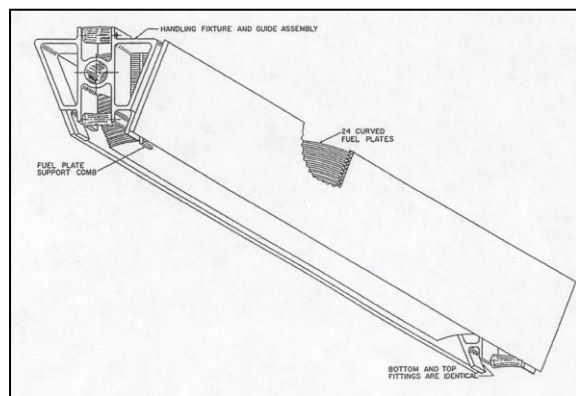

Figure 2
MURR Fuel Element – Pictorial View

The MURR operates continuously with the exception of a weekly scheduled shutdown. Over the past 32 years of operation, the MURR has averaged approximately 6.3 days/week at full power. The weekly shutdown provides an opportunity to access samples in the center test hole, to perform surveillance tests and maintenance, and to replace all eight fuel elements in the core. Replacing the fuel elements provides the chance to remix or shuffle which elements will be used in the core and to restart the reactor with a xenon-free core.

The active fuel cycle typically consists of 32 fuel elements, corresponding to 16 pairs of elements. A core loading will always consist of four (4) different pairs of elements, with the two (2) elements of each pair loaded opposite of each other in the core. The compact core volume limits excess reactivity and causes the control blades to be fully withdrawn when the HEU core, with equilibrium xenon, achieves approximately 670 MWd. This results in an HEU fuel element reaching a maximum burnup of 150 MWd, which in turn corresponds to a hot spot burnup of less than 1.8E+21 fissions/cc. Therefore the Technical Specification limit of 2.3E+21 fissions/cc for the UAl_x dispersion fuel is not approached or exceeded. Cores are usually loaded such that the average power history of a fuel element is a little less than 75 MWd. Typically a fuel element will be used in 18 to 20 different core loadings before being retired from the fuel cycle. A core

Table 1 – Summary of MURR Fuel Element Specifications

Description		Current HEU Fuel	Proposed LEU Fuel
Fuel			
	Material	UAl_x (mostly UAL_3)	U-10Mo
	Enrichment	93% ^{235}U	19.75% ^{235}U
	Thickness Plate-1	20 mil (0.508mm)	9 mil (0.229 mm)
	Plate-2	20 mil (0.508mm)	12 mil (0.305 mm)
	Plate-3 through 23	20 mil (0.508mm)	18 mil (0.457 mm)
	Plate-24	20 mil (0.508mm)	17 mil (0.432 mm)
Cladding			
	Material	Aluminium	
	Thickness Plate-1	15 mil (0.381mm)	20 mil (0.508 mm)
	Plate-2	15 mil (0.381mm)	13 mil (0.330 mm)
	Plate-3 through 23	15 mil (0.381mm)	10 mil (0.254 mm)
	Plate-24	15 mil (0.381mm)	16 mil (0.406 mm)
Fuel Element			
	Number of Fuel Plates	24	
	Overall Fuel Element Length	32.5 inches (82.550 cm)	
	Overall Fuel Plate Length	25.5 inches (64.770 cm)	
	Overall Active Fuel Length	24.0 inches (60.960 cm)	
	Fuel Plate Thickness 1 & 24	50 mil (1.270mm)	49 mil (1.245mm)
	Fuel Plate Thickness 2-23	50 mil (1.270mm)	38 mil (0.965mm)
	Distance Between Plates (Coolant Channel)	80 mil (2.032mm)	92 mil (2.337mm)
	Maximum ^{235}U Loading	775 grams	1439 grams
	Fuel Density	1.53 grams/cm^3	3.03 grams/cm^3
	Weight	~ 6 Kg	~ 11 Kg

with fuel elements approaching the burnup limit will also include a corresponding number of elements with very low power history. This maximizes the number of MWd obtainable per fuel element. This same approach is also planned for the LEU fuel cycle.

3.   LEU FUEL ELEMENT DESIGN

The initial scoping work of MURR's fuel conversion feasibility study included defining the fuel requirements, describing the HEU core, and defining experimental facility performance indicators. Some of the potential concerns in performing a conversion include (1) maintaining the performance and safety characteristics of the current 775-gram ^{235}U fuel element, (2) not increasing the number of fuel elements used per year, and (3) matching or enhancing neutron flux in the center test hole (flux trap), and graphite reflector and beamport regions [4, 5, 6].

To explore the possibilities of an LEU core design that could match or exceed current performance capabilities, the MURR tasked the GTRI Fuel Development Program to answer a number of key questions on the following fuel design/manufacturing limitations: peak burnup, minimum thickness of the fuel meat and cladding, minimum thickness of the curved plate to ensure sufficient rigidness, and the magnitude of engineering peaking factors due to reducing the thickness of the fuel meat. The best information available from U.S. High-Performance Research Reactor Working Group (USHPWG) collaboration during 2006-2009 is:

- What is the peak fuel burnup limit? *~7E+21 fissions/cc*
- How thin can acceptable U-10Mo foils be fabricated? *5 mil (0.127 mm)*

- What is the minimum acceptable cladding thickness? ***10 mil (0.254 mm)***
- How thin can sufficiently rigid curved fuel plates be fabricated? ***38 mil (0.965 mm)***
- Magnitude of engineering peaking factors for thin U-10Mo foils? ***≤ UAl$_x$ HEU factors***
- What is the minimum cladding blister temperature? ***850-900 °F (454-482 °C)***

Note that two of these points have still not been fully confirmed:

- It is not yet clear whether the 10 mil clad thickness will prove too difficult or expensive to fabricate.
- Furthermore, experiments and analyses to prove the hydrodynamic stability of the thin 38 mil fuel plates must still be performed.

Should a thicker clad and/or a stiffer plate be required, then the inherent penalty of displacing moderating water will need to be addressed to prove technical feasibility of an alternate fuel design.

Despite the uncertainty surrounding the cladding thickness, the design work continued based on information indicating that a 10 mil clad should be feasible. Table 1 compares the current HEU fuel element with the proposed LEU fuel element design. To compensate for the reactivity effects associated with conversion to LEU, the water-to-metal ratio was increased by decreasing the plate cladding and fuel meat thickness, while increasing the width of the coolant channel gap. The high peak fuel burnup limit of the monolithic fuel means that the power density peaking factor does not need to be reduced to avoid limiting fuel element lifetime. However, heat flux peaking limits the safe reactor operating power level. Consequently, the heat flux peaking was reduced by thinning the fuel meat, particularly in Plate-1 and -2. Using this approach the proposed core is designed with the fuel meat thickness varying by a factor of two. Most plates have an 18 mil fuel meat; the thinnest fuel meat is 9 mils. All but the outer fuel plates are 38 mils thick. The outer two plates (Plate-1 and -24) are 49 mils thick for the following two reasons: (1) Plate-1 and -24 have the cladding that is most at risk to being scratched or bumped during weekly fuel handling, and (2) these plates also are located between different width coolant channels which can create a differential pressure across them.

4. NEUTRONIC ANALYSES

A joint study between MURR and the GTRI Reactor Conversion Program at Argonne National Laboratory (ANL) was conducted to determine a suitable LEU fuel element design and to perform the preliminary analyses necessary to establish that the shutdown and safety margins remain acceptable for the converted reactor. The collaboration continues in order to finalize the LEU assembly design and perform complete safety analyses for the converted core.

4.1 Description of Neutronics Codes and Methodologies

The codes and methodologies used and developed allowed for tractable yet highly detailed neutronic/depletion calculations. To perform the neutronics calculation of a compact core such as MURR, it is preferable to use a transport theory code to capture the rapidly changing spectra across the various regions. Therefore, the MCNP [7, 8] continuous energy Monte Carlo code
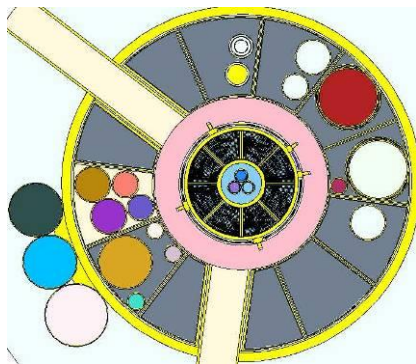
Figure 3
Cross Section of MCNP
Model of MURR

was used for all detailed calculations of core k-effective, control blade position, as well as detailed power distributions and experimental fluxes/reaction rates. Figure 3 illustrates the types of detail in the MCNP model of the current reactor flux trap, core, and reflectors.

REBUS-PC [9] was used for the MURR depletion calculations. The REBUS code is capable of utilizing either diffusion theory or Monte Carlo neutron flux solvers. Although depletion calculations using MCNP for flux and cross section evaluation at each state point can be performed using the REBUS-MCNP [10, 11, 12] computer code, the hundreds of state points required to model the complex fuel cycle of MURR -- a process repeated during fuel element optimization -- made REBUS-MCNP intractable for the full fuel cycle simulation in a reasonable time frame. It was therefore necessary to develop a less time-consuming diffusion model to complete the depletion calculations. The REBUS-DIF3D [13, 14] code was chosen for this portion of the feasibility study. Note that REBUS-MCNP models were still developed for the MURR reactor in order to benchmark the REBUS-DIF3D models. The WIMS-ANL [15, 16] lattice physics code was used to generate a burnup dependent, 69 group lumped fission product to model the fission products not explicitly modeled in the MURR MCNP models [17].

A detailed Theta-R-Z diffusion model was developed for the DIF3D finite difference multigroup diffusion code. WIMS-ANL was used to generate burnup dependent cross section libraries for all the materials of the REBUS-DIF3D model. Considerable customization of the cross-section generation process was required to prepare a robust set of 10 group cross sections. The details of the DIF3D and WIMS-ANL models were developed iteratively to assure fidelity of the resulting diffusion calculations with MCNP and experiments [18].

Since REBUS-DIF3D depletion was only used to model the weekly operation for HEU and LEU while MCNP was used for all neutronics calculations, it was necessary to develop an automated method to update the MCNP models with the detailed three dimensional, burnup dependent material compositions (i.e., atom densities of all the depleting isotopes modeled) obtained from REBUS-DIF3D. Comparisons between DIF3D and MCNP were performed for both the fresh and depleted cores to verify the DIF3D model for this highly complex problem, and to demonstrate that the depleted fuel composition data were correctly transferred from the DIF3D to the MCNP models.

4.2 Development of HEU and LEU Fuel Cycle Models

To compare the performance of the proposed LEU design to typical HEU operation, models were developed for the 2008 reactor configuration with typical experimental loadings. To properly model the current HEU core fuel utilization, thermal-hydraulic safety, and experimental performance, it was necessary to develop a computational shuffling that would accurately model the actual complex cycle used at MURR. It was also necessary to develop a similar shuffling

scheme for the LEU in order to recalculate those parameters and demonstrate that the proposed LEU fuel element described in Section 3 is an acceptable fuel design.

Nine HEU pre-simulation cores were modelled to produce 24 fuel elements (12 pairs) with appropriate power histories ranging from 0 to 139 MWd. Using these, an 82-week simulation of reactor operations with HEU fuel was modeled with the REBUS-DIF3D code. Each week, the simulated reactor loading follows typical loading pattern practices for the MURR. Fresh fuel elements are loaded about every four to five weeks, and fuel elements are discharged from the simulation at the same rate, with a target burnup of 150 MWd. The simulation was conducted for the reactor with current typical reflector and flux trap sample loadings. Additionally, the control blades were fixed at 23 inches withdrawn, which is the typical average blade position during weekly operations, while the regulating blade was positioned at the core mid-plane (13.375 inches withdrawn).

For the purpose of this feasibility study, an LEU fuel cycle simulation with an average EOC burnup of 890 MWd was developed and analyzed. Nine pre-simulation cores with LEU fuel were modelled to produce 12 pairs of fuel elements with appropriate power histories ranging from 0 to 190 MWd. Using these, a 93-week simulation of reactor operations with LEU fuel was modeled with the REBUS-DIF3D code. Fresh fuel elements are loaded about every five weeks in this simulation, and the discharge burnup of the LEU elements is about 208 MWd.

Figure 4 compares the core k-effective from the HEU and LEU fuel cycle simulations at 10 MW and 12 MW, respectively. The k-effectives of the LEU fuel cycle at 12 MW are bounded by the high and low extremes of k-effective for the HEU fuel cycle at 10 MW. The slightly lower average reactivity for LEU at the beginning of the week could be advantageous for experiments, since the control blades would be less inserted. The average EOC core k-effective for the HEU core predicted by the simulation is sub-critical, with k-effective=0.994. However, it was found from comparisons to 1971 critical experiments that the DIF3D model bias is -0.49% Δk/k;
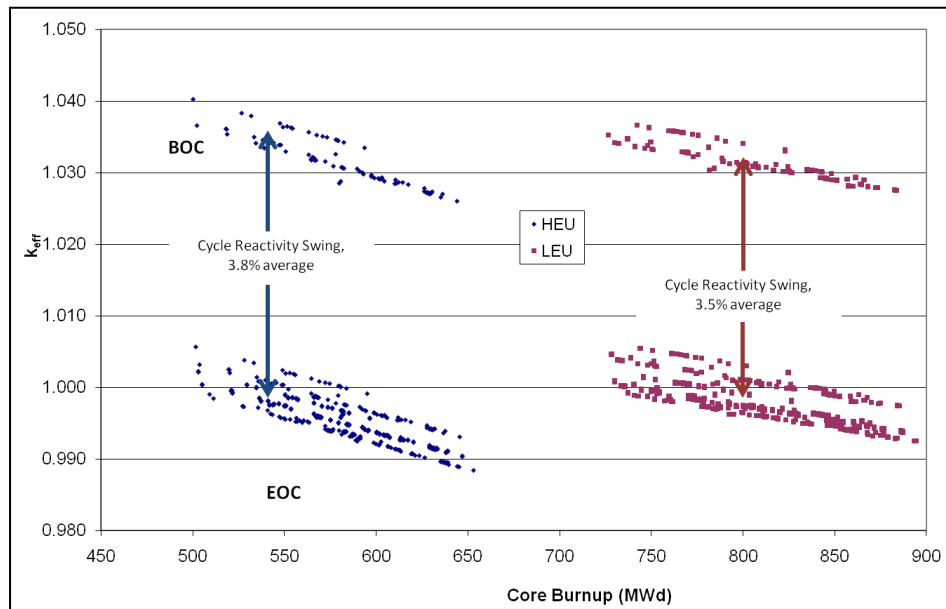

Figure 4 – Weekly Core k-effective for MURR HEU and LEU Fuel Cycle Simulations

7

furthermore, the simulation was performed with the blades fixed at 23 inches withdrawn. The estimated critical positions at BOC and EOC for a number of these cores have been calculated by MCNP and found to be within range of typical values (see Section 4.4). The average weekly reactivity swing for the HEU core is 3.8% Δk/k, while the average LEU weekly reactivity swing is 3.5% Δk/k. The proposed LEU design appears suitable for the weekly fuel cycle at MURR, provided that the element can be fabricated and is demonstrated to be hydrodynamically stable.

4.3 Fuel Cycle Performance

Table 2 provides a comparison of the operating characteristics of MURR with the current HEU fuel and the proposed LEU fuel for the following parameters: maximum burnup, core MWd with the control blades full out, frequency of core refueling, and number of fuel elements in the fuel cycle. Considering the average MURR utilization factor of 90%, one year of operation corresponds to 3,285 MWd. Consequently, the 150 MWd maximum discharge burnup of the HEU elements corresponds to utilization of ≥ 22 elements/year. The LEU fuel cycle developed in Section 4.2 results in an average discharge burnup of 208 MWd. Using the same total annual MWd as the current HEU 10 MW fuel cycle, the number of LEU fuel elements used in a typical year would be 15.8. However, the feasibility study showed that a power increase is required to maintain experimental performance. LEU operation at 12 MW was projected to match HEU experimental performance, or provide small gains. At 12 MW, the proposed LEU fuel cycle would require 18.9 elements/year. This corresponds to a 14% decrease in annual fuel element consumption, despite the power increase.

Table 2 – Current and Proposed MURR Fuel Operating Characteristics

Parameter	Current HEU Fuel	Proposed LEU Fuel
Maximum burnup:	150 MWd/element (1,200 MWd/core) limited by insufficient excess reactivity – this achieves less than 1.8E+21fissions/cc burnup, compared to the Technical Specification limit of 2.3E+21fissions/cc for UAl$_x$ fuel	208 MWd/element (1,664 MWd/core) limited by insufficient excess reactivity – this achieves less than 4E+21 fissions/cc burnup
Core MWd (control blades full out):	~670 MWd core with equilibrium xenon activity (56% of 1,200 MWd)	~890 MWd core with equilibrium xenon activity (53% of 1,664 MWd)
Refuelling:	Weekly – replace all eight fuel elements; fuel elements are used in 18 to 20 core loadings to achieve 145 to 150 MWd burnup at 10 MW (~24% burnup)	Weekly – replace all eight fuel elements; fuel elements are used in ~ 22 core loadings to achieve ~208 MWd burnup at 12 MW (~18% burnup)
Fuel Cycle:	22 elements used per year at 10MW; 32 fuel elements in active fuel cycle	19 elements used per year at 12MW; 32 fuel elements in active fuel cycle

The ability to fabricate plates with 10 mil clad, and the hydrodynamic stability of 38 mil plates must still be demonstrated. The GTRI Fuel Fabrication Capability and Fuel Development programs are addressing both concerns. The modeling capability created to establish feasibility of the current proposed design can be applied in the future to develop a contingency LEU design with thicker plates, and to compare the current and contingency designs with regard to fuel utilization, safety margins, and experimental performance.

With the aforementioned caveat, this analysis shows that, on the basis of number of fuel elements consumed per year, the conversion of MURR using the proposed LEU fuel element and cycle is feasible. Thermal-hydraulic safety margins and experimental performance are demonstrated in Sections 5 and 6.

4.4 Computational Model Credibility – Comparisons with Estimated Critical Positions

No all fresh core has been available since 1980, so no measurements for an all fresh core are available for the 2008 reactor configuration. In order to further demonstrate the credibility of the MCNP 2008 core configuration model and the depletion methodology, a series of 15 Estimated Critical Position (ECP) calculations were performed to determine whether the overall depletion scheme provides a good estimate of typical reactor performance. Each case was a measured critical state at hot conditions. Small differences between the nominal water temperatures and measured temperatures were corrected by applying the reactivity coefficients of HEU operations at MURR. The cores analyzed covered a broad range of fuel loadings (e.g., two fresh elements in the core, or none), a variety of flux trap loading states (including an empty flux trap), and notably, a broad range of control blade history states. Burned material compositions for each element of the critical states were derived from the results of the REBUS-DIF3D fuel cycle simulation to find the closest element average burnup match at a beginning-of-week step.

Table 3 summarizes the cases evaluated, including the deviation of MCNP5 k-effective from 1.0. It is clear that many cases had excellent agreement, but also clear that some cases had a large deviation from critical. The deviations were cross-compared to element burnup, flux trap state,

Table 3 – Summary of Critical States Evaluated for Partially Burned Cores

Date	Fuel Element Burnup (MWd)					Flux Trap Reactivity[1] (%Δk/k)	Ave. Prior Days for Control Blades	Critical Bank Height[2] (inches withdrawn)	MCNP5 Deviation from Critical[3] (%Δk/k)
	X1/X5	X1/X6	X3/X7	X4/X8	Sum				
04/23/05	32	92	73	95	584	0.478	271	17.97	-0.263
05/02/05	38	140	44	73	590	0.474	280	18.02	-0.228
05/09/05	0	117	63	115	590	0.427	287	17.63	-0.260
05/16/05	17	137	52	82	576	0.432	294	17.93	-0.270
05/30/05	9	139	21	124	586	0.474	308	18.06	-0.144
07/11/05	29	136	40	84	578	0.464	350	17.98	-0.257
06/16/00	54	72	41	143	620	0.346	1040	17.22	-1.028
08/07/00	16	98	68	117	598	0.384	1092	17.02	-1.086
11/15/00	0	139	56	108	606	0.359	1192	16.72	-1.065
12/17/01	22	124	69	91	612	0.348	1709	16.64	-1.317
12/31/01	14	131	72	87	608	0.340	1723	16.66	-1.285
04/22/02	0	118	64	114	592	0.418	1835	16.00	-1.697
08/08/05	0	143	38	115	592	0	378	18.52	-0.087
09/04/00	24	90	50	141	610	0	1120	17.81	-0.080
02/04/02	11	136	61	96	608	0	1758	17.03	-0.594

[1] The flux trap reactivity indicates the worth of the flux trap contents relative to an empty flux trap.
[2] Critical bank heights reported here are corrected for small differences between the nominal water temperatures modeled and those measured at the critical state.
[3] MCNP deviation from critical is (k-1)/k, corrected for the difference between flux trap worth of the critical state and flux trap worth modelled with the nominal sample loading (for cases with nonzero flux trap worth).

blade insertion, and prior history of the control blades. The only observable trend was the control blade history.

While the feasibility case is based on fresh control blades in the MCNP model, the MURR control blades are shuffled in a multi-year scheme analogous to fuel shuffling because the Boral material is known to deplete in the region near the tip, which is in a position of high importance. Figure 5 illustrates the clear trend of MCNP deviation from critical vs. the prior history of the control blades. Seven of the 15 ECP cases had little prior use of the control blades: 271-378 average calendar days of prior use. The RMS bias for those seven cases was 0.226% $\Delta k/k$. The trend lines of Figure 5 indicate that if fresh blades were loaded in MURR, the MCNP deviation from critical would be even less.
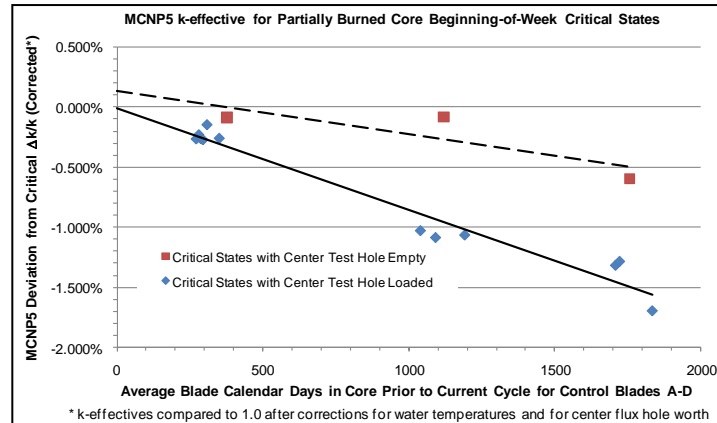

Figure 5
MCNP ECP Deviation from Critical vs. Blade History

5. COMPARISION OF HEU AND LEU SAFETY MARGINS AND PERFORMANCE

5.1 Power Distributions for Steady-State Safety Margin Evaluation

Power peaking in MURR is dependent upon the mix of burnup states among the elements in the core, upon critical control blade positions, and upon the experiment/sample loadings, particularly in the flux trap. The mix of burnup states of the elements within the core largely determines the power sharing between elements. A core with both fresh elements and elements that will be discharged at the end of the cycle is expected to have the highest element peaking factor since the fresh elements must produce more power than they would in an all fresh core. However, the critical control blade position is higher in the mixed core than in an all fresh core.

The critical control blade position is important to the power distribution in two ways. First, a lower blade suppresses power in the outer plate(s) (since the MURR control blades are radially adjacent to the outer fuel plates) and therefore increases power in the inner plates. The radial shift effect is important for margin to flow instability since it results in a change in cooling channel temperature rise. The lower blade also increases axial peaking, particularly in the outer plates. The change in axial peaking is not important for margin to flow instability, but this change is important to critical heat flux since the local axial clad temperature is proportional to the local heat flux. It appears that the safety limits will only be limited by flow instability, but critical heat flux analysis will be included in the detailed safety analysis to verify this.

As equilibrium xenon builds up during operation, the control blades are withdrawn to compensate for the negative reactivity of the xenon. The change in control blade position shifts the power, as discussed above. The xenon may also alter the power shape directly since it builds

up in the regions of highest power. The balance of outward power shift due to blade withdrawal and inherent power flattening by local xenon buildup should be explicitly evaluated.

The experiment/sample loading is also important to power distributions in two ways. First, the displacement of water in the flux trap region from samples loaded results in a reactivity insertion due to its positive void coefficient. Therefore, loading of samples in the flux trap leads to an increase in core reactivity, and thus lower critical blade position. The displacement of moderator from the flux trap or an absorption in the samples could have a direct effect on the power in the interior plates. But the lower control blades would tend to push power inboard. Sample loading in the graphite reflector positions also has a small effect on core reactivity, and thus on the critical control blade position. The balance of the effects must be explicitly evaluated.

After considering the various contributors to power peaking discussed above, power distributions were calculated for 16 cases that enveloped the distinct combinations of effects. For the HEU core, the two highest heat fluxes were in either a core of eight (8) new fuel elements (labeled Case 1 in the following discussion) or the Week 58 core from the fuel cycle simulation (Case 3), which had four (4) pairs of fuel elements with the following power history: 0, 81, 65, and 142 MWd. For the proposed LEU core, the two highest heat fluxes were in either a core of eight (8) new fuel elements (Case 5) or the Week 79 core of the simulation (Case 7), which had four (4) pairs of fuel elements with the following power history: 0, 116, 97, and 199 MWd. Initial startup with no xenon was compared to the equilibrium xenon state at two days of operation. Furthermore, a typical flux trap loading was compared to an empty flux trap. Thus eight cases were considered for both HEU and the proposed LEU design.

The atom densities of the fuel compositions for each core state were read from REBUS-DIF3D depletion results to update an MCNP input file. An automated search was then performed with MCNP to find the critical banked blade position for the core (i.e., blades moved until MCNP predicted a k-effective of 1.0). Detailed power distributions were calculated with MCNP by tallying the fission power (f7 tally) within 24 radial (i.e., plate-by-plate), 24 axial, and 9 azimuthal segments of the fuel plate meat in the entire core of eight elements (i.e., 216 equal volume segments within each plate; 5,184 segments per element). Finally, a post-processor was applied to read the MCNP *mctal* file and produce normalized edits suitable for analysis (and to facilitate an automated linkage to the PLTEMP/ANL thermal hydraulics code discussed in Section 5.2). It should be noted that credit for power deposition outside the fuel is not modeled here, but is taken into account in the thermal-hydraulic safety margin calculation.

Figures 6 and 7 show the axial heat flux profiles for each plate in element position X1 in the two most limiting cases for flow instability in the HEU and LEU cores, respectively. The impact of control blades is evident. The outer plate heat fluxes are significantly skewed toward the bottom of the core, below the tip of the critical blade positions. The axial (plate-by-plate) shapes of plate power could be compared as either power density (W/cm^3) or heat flux (W/cm^2). However, heat flux on the plate surface is the appropriate quantity to compare axially along the fuel plates in order to predict heat transfer to the coolant. Furthermore, power density is not a good parameter to compare due to variations in the fuel meat thicknesses; the fuel meat volumes per plate are different between LEU and HEU, and for the proposed LEU fuel, the three highest power density plates have thinner fuel foils than the rest to reduce their heat flux.
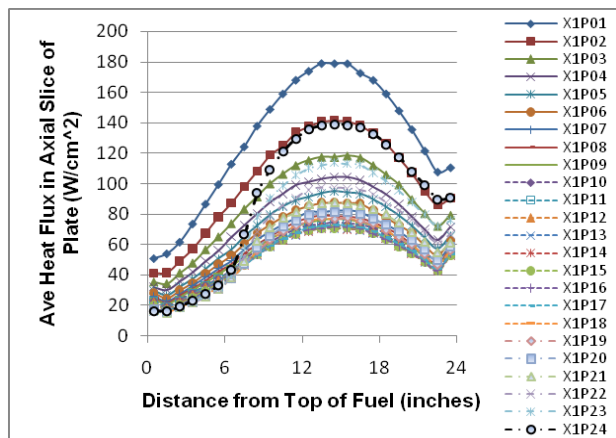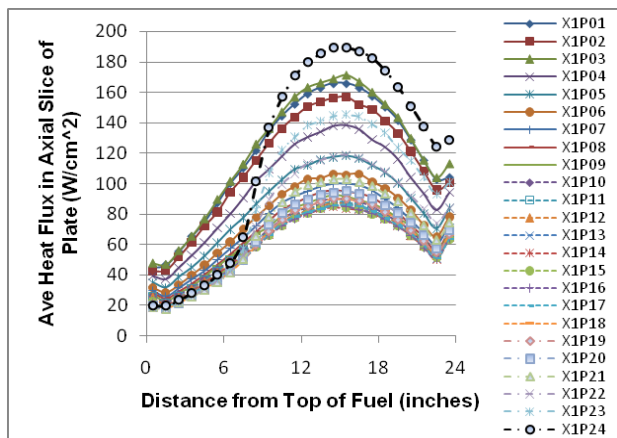
Figure 6 – HEU Core 3B Element X1

Figure 7 – LEU Core 7B Element X1

The margin to flow instability is primarily dependent upon total heat transferred to a coolant channel rather than the axial shape of the heat flux. Figure 8 illustrates the axial average heat flux for each plate of each element in Case 7B. The peaking of the outer plates is not dominant for margin to flow instability, since the presence of the control blade suppresses the total power produced by the outer plates. The peak axial average heat fluxes for Case 7B are in Plate-1 and Plate-3 of element position X1. Plate-1 generally has a high heat flux due to the moderator of the flux trap region -- particularly for the case with no flux trap samples loaded. But the proposed LEU design has a thinner fuel foil (9 mil thick) in Plate-1 to reduce the peaking. The foil in Plate-2 is 12 mil thick, and Plate-3 is the first plate with an 18 mil fuel foil, which explains why the Plate-3 heat flux is slightly higher than the Plate-1 heat flux. The radial shape of the heat flux in the figure also illustrates the important effect of moderation and fissile material self-shielding. The inner and outer plates have a much higher heat flux (i.e., fission rate) due to their proximity to the heavily-moderated flux trap (Plate-1) and reflectors (Plate-24). The interior plates have a lower heat flux due to both less moderation from the coolant channels and the self-shielding effect of the outboard plates consuming thermal neutrons.

The effects of moderation and self-shielding are also important in the azimuthal direction of the MURR elements since there is an unfueled region of the plates adjacent to the side plates. Thus, the fuel near the side plates is in a region with more moderator and less self-shielding than the azimuthal interior region. The MCNP fission power was tallied for nine (9) azimuthal stripes of equal
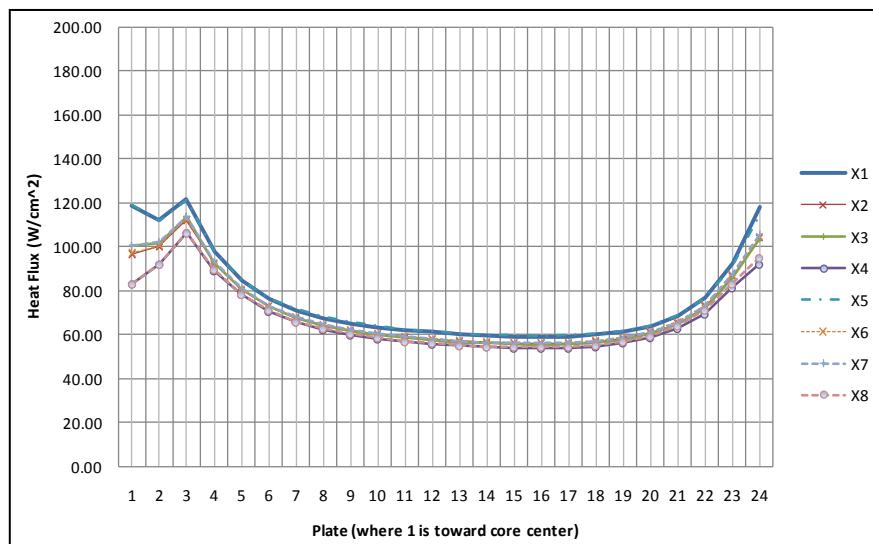

Figure 8 – Axial Average Heat Flux in Each Plate of Each Element
Case 7B: LEU Week 79 Day 0 with Empty Flux Trap

angular span within the fuelled region of the plates. Figure 9 plots the azimuthal peaking factor for each plate of each element for Case 7B. The effect is small (~5%) for the outboard plates since there is already significant moderator and little self-shielding, but clearly pronounced for the interior plates (15-23%). Fortunately, the average heat flux in the interior plates is lower than in the outboard plates, so
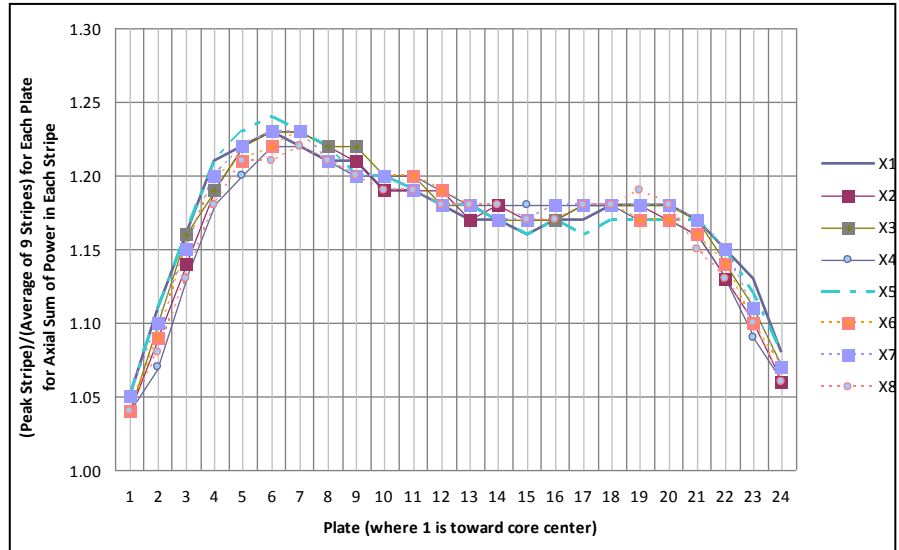

Figure 9 – Azimuthal Peaking Factor for Each Plate of Each Element
Case 7B: LEU Week 79 Day 0 with Empty Flux Trap

the largest azimuthal peaking factors do not correspond to a "hot stripe" for the entire element.

The "hot stripe" heat flux can be thought of as the multiplication of plate average heat flux (Figure 8) times the plate azimuthal peaking factor (Figure 9), and is an effective figure of merit to compare probable effects of the core power distributions on the margin to flow instability. The 'hot stripe' heat fluxes for each plate of each element were tallied explicitly. Table 4 summarizes the 'hot stripe' heat fluxes in the key plates of the fresh fuel elements for each of the 16 cases for which power distributions were evaluated. Cases 3B and 7B are for cores without xenon, so the lower critical control blade position leads to high inboard heat flux. Cases 4B and 8B are for the equilibrium xenon state of the same core, so the higher critical control blade position allows the power to shift toward the outboard plates. Thermal hydraulic analyses were performed for those four cases (using full power distribution detail, not just the summary of Table 4), as described in Section 5.2.

5.2   Thermal Hydraulic Results

The PLTEMP/ANL code [19] is capable of modeling all of the MURR fuel elements and considering all of the fuel plates and coolant channels of each element simultaneously. The code includes a hot channel factor on bulk coolant temperature rise, accounting for uncertainties in parameters such as calculated power distribution, flow distribution, fuel plate loading, and coolant channel thickness. The code also has a search capability which automatically adjusts the reactor power level until a minimum specified value of the flow instability ratio is achieved.

The predicted margin to flow instability was the criterion used in qualifying the current HEU core [20, 21]. Figure 10 shows the reactor power at which flow instability is predicted to occur in each channel of elements 1 and 5 of the HEU and LEU cores for the reference cores evaluated. The flow instability power level was calculated at the LSSS values for coolant pressure (75 psia at the pressurizer), temperature (155 $^{\circ}$F at the core inlet), and flow (3200 gpm through the core).

Table 4 – Summary of Key Hot Stripe Heat Fluxes Evaluated

Core State that may bound power peaking					Hot Stripe Heat Flux (W/cm^2) Fresh Element in Position X1				Hot Stripe Heat Flux (W/cm^2) Fresh Element in Position X5			
Fuel[1]	Case	Burnup State	Day	Flux Trap[2]	Plate 1	Plate 3	Plate 23	Plate 24	Plate 1	Plate 3	Plate 23	Plate 24
HEU 10 MW	1A	Fresh	0	Samples	126.7	91.4	67.3	76.8	128.8	94.0	69.4	80.4
	2A	Fresh	2	Samples	121.6	89.3	74.4	87.3	123.4	89.4	74.8	86.6
	3A	Week 58	0	Samples	131.7	96.6	82.6	96.6	132.3	97.6	79.3	91.8
	4A	Week 58	2	Samples	126.3	92.6	90.4	107.4	125.6	92.6	82.8	97.8
	1B	Fresh	0	Empty	133.2	94.5	66.7	77.2	133.8	96.2	70.0	80.2
	2B	Fresh	2	Empty	127.0	91.3	74.5	87.9	129.3	92.1	74.3	87.1
	3B	**Week 58**	**0**	**Empty**	**138.6**	**99.3**	**83.0**	**97.6**	**138.9**	**99.7**	**78.9**	**92.2**
	4B	**Week 58**	**2**	**Empty**	**132.9**	**94.8**	**90.8**	**109.6**	**132.1**	**93.2**	**82.8**	**97.9**
LEU 12 MW	5A	Fresh	0	Samples	116.3	134.4	84.9	100.0	119.4	136.6	90.1	107.0
	6A	Fresh	2	Samples	112.2	129.5	94.6	116.0	113.4	130.4	95.8	117.2
	7A	Week 79	0	Samples	119.0	137.6	103.3	126.6	118.4	137.7	101.3	122.3
	8A	Week 79	2	Samples	114.1	130.4	113.8	142.6	113.3	130.1	105.5	131.1
	5B	Fresh	0	Empty	124.0	139.0	85.0	100.8	125.3	140.9	90.8	108.0
	6B	Fresh	2	Empty	119.1	132.4	95.8	118.0	119.6	133.1	96.4	118.2
	7B	**Week 79**	**0**	**Empty**	**124.9**	**141.0**	**104.7**	**127.6**	**125.1**	**140.8**	**102.0**	**123.2**
	8B	**Week 79**	**2**	**Empty**	**120.3**	**133.9**	**114.3**	**145.4**	**119.4**	**132.8**	**105.7**	**131.3**

[1] Note that HEU operates at 10 MW, while 12 MW is proposed for LEU operation. Thus a 20% increase in LEU heat flux would be expected if the element was not altered (in design and underlying physics).
[2] Samples indicates a typical loading of samples in all three flux trap tubes. Empty indicates neither samples nor tubes in the flux trap (i.e., "empty island" configuration).
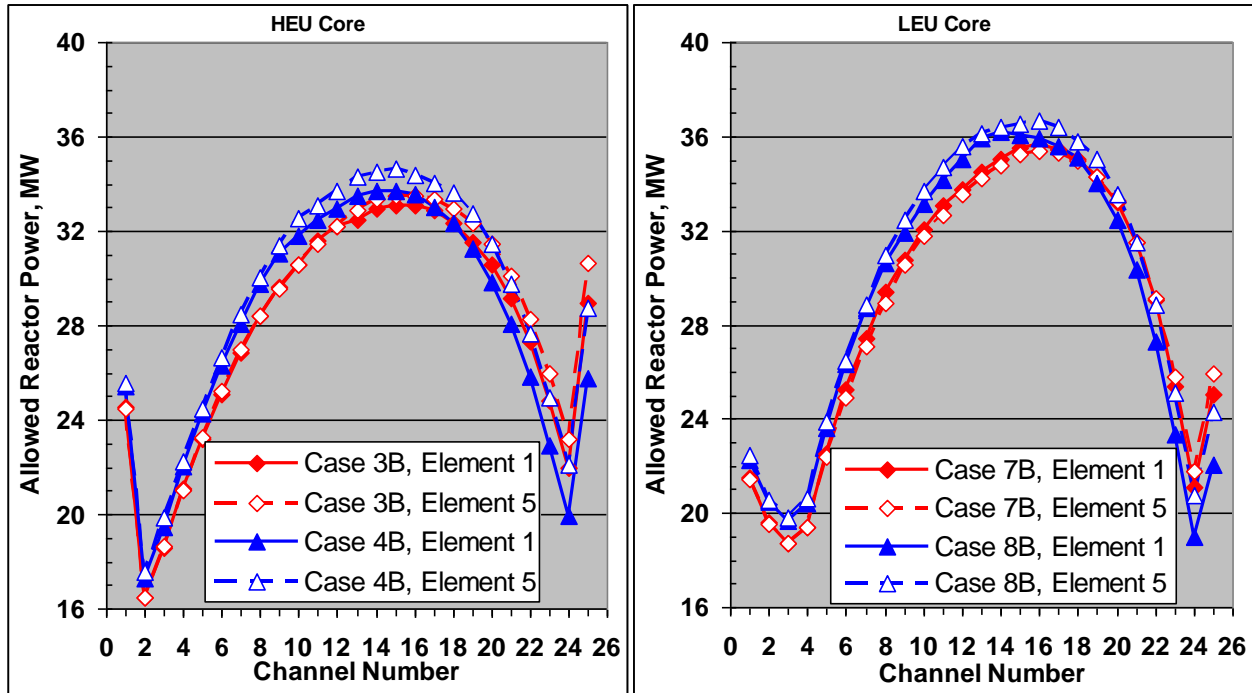

Figure 10 – Reactor Power Predicted to Initiate Channel Flow Instability in Each Core

The distributions of channel flow instability power shown in Figure 10, with the lowest allowed powers near the inner and outer plates of the elements and the highest allowed powers towards the middle, are consistent with the shape of the axial average heat flux per fuel plate illustrated in Figure 8. As is expected, the values of allowed power tend to be at their lowest values where the heat fluxes are at their highest.

Table 5 summarizes the minimum value of the flow instability power in each of the eight plots of Figure 10. The minimum HEU power for flow instability is 16.5 MW. The minimum occurs in channel 2 of element 1 in Case 3B. For this case, channel 2 of element 5 is nearly equally limiting. Similarly, the minimum LEU power is 18.7 MW. It occurs in channel 3 of element 1 in Case 7B. For this case, channel 3 of element 5 is nearly equally limiting as is channel 24 of element 1 in Case 8B. For the HEU core, the 10.0 MW nominal value versus the predicted allowed power of 16.5 MW implies a 6.5 MW margin. This is to be compared with a 6.7 MW margin predicted for the LEU core (16.7 MW versus 12.0 MW).

Table 5 – Flow Instability Power

Case	Element	Power, MW	Channel
HEU			
3B	1	**16.48**	2
	5	16.51	2
4B	1	17.30	2
	5	17.58	2
LEU			
7B	1	**18.73**	3
	5	18.74	3
8B	1	18.98	24
	5	19.79	3

References 20 and 21 predict values for the onset of flow instability based on the original HEU analysis. This analysis used power peaking factors for the HEU core that are more conservative than the 3-dimensional values derived from the MCNP modeling of core power peaking factors. This is due to the power peaking factors that were obtained in 1973 from combining two different sets of peaking factors from two different 2-dimensional EXTERIMANATOR code models. The radial and axial peaking factors from an R/Z model of all fresh fuel elements were combined with the azimuthal peaking factor from a theta/R model of fresh and highly burned up fuel elements combined. This produces an overall peaking factor that exceeds the actual overall 3-dimensional peaking factor for a worst case core. These references provide digital values for 75 psia at the pressurizer and 3200 gpm flow for reactor inlet temperatures of 140 and 160 °F. Linear interpolation of these two flow instability values yields 14.9 MW for a reactor coolant inlet temperature of 155 °F. Since the nominal reactor power for the HEU core is 10.0 MW, in the original analysis a 4.9 MW margin (14.9 MW versus 10.0 MW) was deemed acceptable. Since the nominal reactor power for the LEU core is 12.0 MW and the flow instability power in the current analysis of the LEU core is 18.7 MW, the margin is even larger, 6.7 MW.

In conclusion, based on the sole steady-state thermal-hydraulic requirement established in the original HEU safety analysis [20, 21] that flow instability be avoided, the performance of the proposed LEU core is acceptable.

6. EXPERIMENTAL PERFORMANCE – FLUXES AND REACTION RATES AT IRRADIATION LOCATIONS

The conversion of the MURR from HEU to LEU fuel will affect experimental fluxes. The effects were examined by calculating flux and reaction rate predictions in a number of important experimental locations for several core states.

Based on current and projected MURR utilization, three major experimental regions were selected for comparing the effect of an HEU to LEU fuel conversion. A summary of the many comparisons made are given in Table 6. The tally runs were performed with 600M histories, resulting in the RMS relative error noted in the table.

Table 6 – Comparison of Day 2 LEU Fluxes and Reaction Rates to HEU[1]

Metric	Neutron Energy Range					
	≤ 1 eV		> 1 eV		Sum	
LEU 10 MW, Week 79, Day 2 - Critical Bank 23.481inches withdrawn, Regulating Blade 15 inches withdrawn						
S-32 (n,p) reactions[2] in FT Tube B 13-15 inches[3]	n/a	n/a	95%	± 0.3%	95%	± 0.3%
Flux[4] in FT Tube B 13-15 inches	87%	± 0.1%	93%	± 0.1%		
Ir-191 (n, γ) reactions in FT Tube C 17-20 inches	87%	± 0.3%	93%	± 1.5%	87%	± 0.3%
Ir-193 (n,γ) reactions in FT Tube C 17-20 inches	87%	± 0.3%	88%	± 1.9%	87%	± 0.5%
Flux in Ir wires of FT Tube C 17-20 inches	87%	± 0.3%	92%	± 0.4%		
Flux in Wedge No. 3 Row 1 P-Tube Bottom 3 inches	86%	± 0.1%	91%	± 0.2%		
Si-30 (n,γ) reactions in Green-5 Sample Position	88%	± 0.0%	90%	± 1.4%	88%	± 0.1%
Flux in Green-5 Sample Position	88%	± 0.0%	91%	± 0.1%		
LEU 12 MW, Week 79, Day 2 - Critical Bank 23.481inches withdrawn, Regulating Blade 15 inches withdrawn						
S-32 (n,p) reactions in FT Tube B 13-15 inches	n/a	n/a	113%	± 0.3%	113%	± 0.3%
Flux in FT Tube B 13-15 inches	104%	± 0.1%	112%	± 0.1%		
Ir-191 (n,γ) reactions in FT Tube C 17-20 inches	104%	± 0.4%	112%	± 1.8%	105%	± 0.4%
Ir-193 (n,γ) reactions in FT Tube C 17-20 inches	104%	± 0.4%	106%	± 2.3%	105%	± 0.7%
Flux in Ir wires of FT Tube C 17-20 inches	105%	± 0.4%	110%	± 0.5%		
Flux in Wedge No. 3 Row 1 P-Tube Bottom 3 inches	104%	± 0.1%	110%	± 0.2%		
Si-30 (n,γ) reactions in Green-5 Sample Position	105%	± 0.0%	108%	± 1.7%	105%	± 0.2%
Flux in Green-5 Sample Position	105%	± 0.0%	109%	± 0.1%		

[1] Compared to: HEU 10 MW, Week 58, Day 2, Critical Bank 24.031inches withdrawn, Reg Blade 15 inches withdrawn.
[2] Reaction rates were compared as Reactions/s.
[3] Axial positions noted as inches above bottom of flux trap sample holder.
[4] Fluxes were compared as n/s/cm².

The Center Test Hole or Flux Trap (FT) irradiation positions support one of MURR's primary missions in producing high specific activity isotopes for various applications, including medical use. While the majority of isotopes are produced through thermal neutron reactions, a few require fast neutron threshold reactions. Detailed 2 group and 69 group tallies in MCNP were used to compare HEU and LEU values of:

- Flux and S-32 (n,p) P-32 reaction rate in FT tube B, 13-15 inches above bottom of holder;

- Flux and reaction rates of Ir-191 (n,γ) and Ir-193 (n,γ) in FT tube C, 17-20 inches above bottom of holder, where Ir wires were explicitly modeled in an aluminum holder; and
- Flux in FT tube B, 6-8 inches above bottom of holder.

The Graphite Reflector Region irradiation positions are used to irradiate various sample materials. Detailed 2 group and 69 group tallies in MCNP were used to compare HEU and LEU values of:

- Flux in the bottom three inches of Graphite Wedge No. 3, Row 1 Pneumatic Tube; and
- Flux and Si-30 (n,γ) reaction rate in the Green-5 position, where a 5-inch silicon sample was modeled.

There are four (4) radial and two (2) radial-tangential beamports which have supported major advances in material sciences research. A new facility has also been installed to provide for testing new boron compounds for BNCT in small animals. Detailed tallies in MCNP were used to compare HEU and LEU values for Beamport 'E.' The primary use of beamport research involves thermal flux below 0.1 eV, which would be reduced to 86% of the HEU core flux level by using LEU at 10 MW, but would increase to 104% of the HEU results with LEU at 12 MW.

The fluxes and reaction rates in Table 6 were calculated for reference cores described in Section 5.1. The cores were Week 58 of the HEU simulation and Week 79 of the LEU simulation. Both the beginning-of-week core at Day 0 (i.e., no Xe, lower control blades) and the same core depleted to Day 2 (equilibrium Xe, higher control blades) were examined. Only the results for Day 2 are given in the table.

All tallies were normalized by post-process to allow different power levels to be compared. While the LEU depletions were performed at 12 MW, it would be possible to define an LEU fuel cycle for 10 MW operation. Since the fresh and most depleted elements would not have significantly different burnup for a 10 MW LEU fuel cycle, we assume that the overall power sharing and associated flux profile would not be significantly different.

It is clear from the tables and figures that the flux and reaction rate losses would exceed 10% if the power level of 10 MW is maintained for LEU operation. An uprate to 12 MW would provide modest benefit for all of the fluxes and reaction rates tallied. This analysis also justifies a more in depth look into the thermal safety margins of the LEU core at 12 MW.

On the basis of experimental performance, the conversion of MURR using the proposed LEU fuel element is feasible at 12 MW.

7. FEASIBILITY STUDY CONCLUSIONS AND RECOMMENDATIONS

The analyses performed show that the MURR reactor can be operated safely with the new LEU fuel elements, if the U-10Mo monolithic fuel can be qualified and manufactured.

As has always been true for reactor conversion projects, full safety analyses need to be performed and regulatory approvals received before the reactor will be able to convert to LEU.

It is important to note that the U-10Mo Monolithic Fuel is not yet qualified or commercially available. The Fuel Development (FD) and Fuel Fabrication Capability (FFC) efforts within the GTRI Reactor Conversion Program are both working to clarify the fuel specifications that will be supported for the new LEU fuel. The positive feasibility results reported at this time are predicated on the best information available to date, as communicated through the U.S. High Performance Research Reactor Working Group (USHPWG).

Reactivity to maintain operating lifetime with LEU has been obtained by increasing the water-to-metal ratio in the core by thinning the fuel plates and increasing the coolant channel widths. Plate-2 through -23 were decreased from 50 mil thick to 38 mil by designing for 18 mil U-10Mo foil with 10 mil clad (including any interlayer to control fuel swelling behavior). Reduced foil thicknesses have been designed in three (3) of the fuel plates in order to control power peaking and assure safety margins. It is not yet clear whether the 10 mil clad thickness will prove too difficult or expensive to fabricate. Furthermore, experiments and analyses to prove the hydrodynamic stability of the thin 38 mil fuel plates must still be performed. Should a thicker clad and/or a stiffer plate be required, then the inherent penalty of displacing moderating water will need to be addressed to prove technical feasibility of an alternate fuel design. A contingency design for thicker fuel plates will be explored in the year ahead, in parallel with safety analyses of the current LEU design.

Furthermore, acceptable experimental fluxes will only be maintained if the reactor power can be increased from 10 MW in order to offset the inherent penalty of introducing ^{238}U into the core. Feasibility studies to date have indicated that safety margins will be maintained with LEU fuel operated at 12 MW. The power uprate will be modeled in safety analyses. Regulatory issues of the uprate must be addressed in the near term to assure successful conversion on the aggressive GTRI schedule.

Finally, we must also note that the economic feasibility of conversion cannot be declared until commercial availability of the fuel has been developed, including credible fuel costs projections. MURR understands that GTRI is committed to addressing fuel cost differentials associated with conversion from HEU to LEU. MURR will continue to work within the USHPWG to assist the FFC in development of cost models and/or to pursue redesigns (as possible) once key cost factors are better understood.

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